February 17, 2004

MEMORANDUM TO: Darrell A. Roberts, Acting Chief
Section I-2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Robert Dennig, Chief /RA/
Containment and Accident Dose Assessment Section
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: RESPONSE TO DECEMBER 8, 2003, LETTER FROM THE STATE OF VERMONT RELATED TO THE PROPOSED POWER UPRATE OF THE VERMONT YANKEE NUCLEAR POWER STATION ITEMS 2a1, 2a2, 2a3, 2b, 2c, AND 2d (TAC NO. MC1505)

The State of Vermont, Department of Public Service, in a letter dated December 8, 2003, requested that the NRC respond to questions regarding the proposed 20% power uprate for the Vermont Yankee Nuclear Power Station. The Probabilistic Safety Assessment Branch (SPSB) was requested, by an NRR work request dated December 24, 2003, to respond to questions 2a1, 2a2, 2a3, 2b, 2c, and 2d. Attached are our proposed responses.

Question 2a2 questions the necessity of permitting credit for containment accident pressure in determining available net positive suction head for emergency core cooling system (ECCS) pumps. SPSB considers this a policy issue and not a technical issue and we therefore do not provide a response. This has been discussed with the project manager. With this exception, our proposed responses are attached.

The SPSB review is complete.

cc: R. Ennis

CONTACT: Richard Lobel, SPSB/DSSA/NRR
415-2865

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Attachment: As stated

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OFFICE	SPSB	SPSB:SC
NAME	RLobel:nxh2	RDennig
DATE	02/12/04	02/17/04

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PROBABILISTIC SAFETY ASSESSMENT BRANCH

RESPONSE TO

DECEMBER 8, 2003, LETTER FROM

THE STATE OF VERMONT

RELATED TO THE PROPOSED POWER UPRATE OF THE

VERMONT YANKEE NUCLEAR POWER STATION

ITEMS 2a1, 2a2, 2a3, 2b, 2c, AND 2d

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The original question is restated in italics followed by the Probabilistic Safety Assessment Branch's (SPSB's) proposed response.

2. *We have questions regarding Entergy's request to change its licensing basis to allow crediting of containment pressure for calculating certain pumps [sic] net positive suction head (NPSH) following postulated loss-of-coolant accidents (LOCA), station blackout, and Appendix R fire events:*

a. *It appears the base guidance for reviewing this area is Standard Review Plan (SRP) 6.2.2, Containment Heat Removal Systems, Rev.4, October 1985. SRP 6.2.2 appears to follow Regulatory Guide 1.1 (Safety Guide 1) and is unequivocal that credit may not be taken for containment pressurization for NPSH considerations. However, the draft Review Standard for Extended Power Uprates, RS-001, December 2002, indicates that the review standard for this area is SRP 6.2.2, as supplemented by Draft Regulatory Guide (DG) 1107, Water Sources for long-term Recirculation Cooling following a Loss-of-Coolant Accident, February 2003. DG 1107, at 7, includes the statement:*

ATTACHMENT

> *Predicted performance of the emergency core cooling and the containment heat removal pumps should be independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions...However, for some operating reactors, credit for containment pressure may be necessary. This should be minimized to the extent possible.*

1) What guidance does the agency have for determining whether "credit for containment accident pressure [is] necessary"?

RESPONSE:

At the time of this response, the Entergy request to increase the Vermont Yankee Nuclear Power Station rated thermal power by 20% is under review. The responses to the questions of the state of Vermont are therefore answered in general terms. The NRC has not reached any conclusions concerning the licensee's request at this point in the review.

The NRC has allowed credit for a portion of the calculated containment accident pressure in calculating the available net positive suction head (NPSH) of the emergency core cooling system and containment heat removal pumps in some boiling water reactors (BWRs) and, in fewer cases, in pressurized water reactors (PWRs). Such credit is allowed when: (1) there is no other practicable way to maintain the available NPSH greater than the required NPSH during postulated design basis accidents, (2) analysis has demonstrated that, using conservative assumptions, this pressure will be available for the design basis accidents, and (3) when examined from a broader perspective than just the design basis accidents, an acceptable level of safety is still maintained.

If credit for containment accident pressure were not given, the alternatives would be, depending on the situation:

> Reducing the reactor power when adequate available NPSH cannot be demonstrated otherwise at the current power level,
>
> remaining at the current power level when an increase in licensed power level (power uprate) is otherwise justified, or
>
> installing new ECCS or containment heat removal pumps or other major modifications

The NRC considers these options to be unnecessary when, as stated above, evaluation and analysis have demonstrated that the credited pressure will be available for the design basis accidents, and when examined from a broader safety perspective than just the design basis accidents, an acceptable level of safety is maintained.

In order to understand the NRC's position, we consider it helpful to consider the evolution of this position.

Regulatory Guide 1.1 (Safety Guide 1), issued in November 1970, requires that

> Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated loss-of-coolant accidents.

Reactors licensed after issuance of Regulatory Guide 1.1 generally complied with this guidance.

On December 3, 1985, the NRC issued Generic Letter 85-22[1] which discussed the findings of the NRC Unresolved Safety Issue (USI) A-43,"Containment Emergency Sump Performance." This issue concerned the blockage of emergency core cooling system sump screens in PWRs. The technical findings of USI A-43 are documented in NRC report NUREG-0897 Revision 1[2]. GL 85-22 discussed these findings which included the fact that (1) blockage of sump screens by LOCA-generated debris requires a plant-specific resolution, and (2) a revised screen blockage model should be applied to emergency sump screens. However, the NRC regulatory analysis of this issue resulted in the decision to not backfit these findings. GL 85-22 recommended that the technical guidance developed while studying this issue be used by licensees for any future modifications to thermal insulation inside containment or to primary coolant system piping.

As part of the resolution of USI A-43, Standard Review Plan Section 6.2.2 was revised (Revision 4) to include the following guidance on NPSH:

> The NPSH analysis will be acceptable if (1) it is done in accordance to the guidelines of NUREG-0897 and (2) it is done in accordance with the guidelines of Regulatory Guide 1.1.

Thus, even after this first examination of the effects of LOCA-generated debris on the available NPSH of ECCS pumps, the criterion for calculating available NPSH remained that of Regulatory Guide 1.1.

On July 28, 1992, the Barsebäck Unit 2 boiling water reactor in Sweden experienced a spurious opening of a pilot-operated relief valve at 435 psig which resulting in dislodging mineral wool insulation which subsequently blocked emergency pump suction strainers. This subsequently led to the NRC issuing NRC Bulletin 96-03[3]. All BWRs complied with the recommendations of Bulletin 96-03, including Vermont Yankee, by the installation of larger, better designed ECCS

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[1] NRC Generic letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985

[2] <u>Containment Emergency Sump Performance, Technical Findings related to Unresolved Safety Issue A-43</u>, NUREG-0897 Revision 1 US NRC October 1985

[3] NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," May 6, 1996

suction strainers. The design of these strainers took into account plant-specific suction strainer loadings of several types of materials including LOCA-generated debris from dislodged thermal insulation, dislodged paint chips and rust accumulated in the suppression pool which would become thoroughly mixed in the suppression pool water by the turbulence generated by the LOCA. In general, these loadings were predicted to be much higher than anticipated prior to the research which followed the Barsebäck event. This resulted in an increase in the predicted flow resistance across the strainers which resulted in a decrease in calculated available NPSH. In some cases this necessitated credit for containment accident pressure. This was not true for Vermont Yankee. The improved suction strainers were installed in Vermont Yankee during the 1998 refueling outage[4]

As a related issue, in 1996 and 1997, as a result of NRC inspections, licensee notifications, and licensee event reports, the NRC staff became aware that the available NPSH for ECCS and containment heat removal system pumps may not have been adequate in all cases. This applied to both PWRs and BWRs. In order to understand the extent of the problem, the NRC issued Generic Letter 97-04[5] requesting licensees to provide current information regarding NPSH analyses for the ECCS and containment heat removal pumps. In some cases, in response to GL 97-04, licensees revised their NPSH analyses and in some of these cases licensees proposed credit for containment accident pressure in the calculation of NPSH. This was necessitated by several factors: calculations that incorrectly omitted an important effect (such as underestimating flow losses), an increase in estimated debris loading on BWR ECCS suction strainers in response to NRC Bulletin 96-03, or an increase in suppression pool temperature (due to degradation of the heat transfer capability of the heat exchangers in the suppression pool cooling system). The NRC reviewed all responses to GL 97-04. In some cases, especially those in which credit was taken for containment accident pressure, the NRC performed detailed reviews. The NRC staff formulated and applied acceptance criteria for these reviews. These criteria were not documented in a publically available source at that time. In order to document these criteria for future use, and to make them available to stakeholders, the NRC staff included them in Draft Regulatory Guide (DG) 1107. Including regulatory positions on NPSH in this draft regulatory guide provided one reference for all regulatory positions related to pump suction issues (vortexing, air entrainment, debris blockage as well as NPSH). DG-1107 was finalized and published as Regulatory Guide 1.82 Revision 3 in November 2003.

There are no review criteria in GL 97-04 itself. GL 97-04 was a request for information. Specifically, there was no criterion prohibiting the use of containment accident pressure in the calculation of available NPSH in GL 97-04.

The NRC staff briefed the Advisory Committee on Reactor Safeguards (ACRS) twice on the calculation of NPSH and credit for containment accident pressure. The last briefing, on December 3, 1997, particularly concerned the staff's position on credit for containment accident

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4 Letter from Don M. Leach, Vice President-Engineering, to the US Nuclear Regulatory Commission, December 29, 1999.

5 Generic Letter 97-04, "Assurance of Net Positive Suction Head for Emergency Core cooling and Containment Heat Removal Pumps," US Nuclear Regulatory Commission, October 7, 1997

pressure in determining available NPSH during beyond design basis accidents. A December 12, 1997, ACRS letter to the NRC Chairman, Shirley Ann Jackson, concurred in the NRC staff position but urged that all accident sequences should be examined. This aspect is discussed further below.

Credit for containment accident pressure is allowed only for the situation in which an existing ECCS or containment heat removal pump cannot otherwise demonstrate sufficient available NPSH. The NRC has made the judgment that, in these cases, the impact of replacing existing ECCS or containment heat removal pumps with pumps that do not require this credit, or taking one of the alternative actions listed above, is not justified based on the design basis safety analyses done by each plant.

This judgment is based on several factors. First, only a fraction of the total containment accident pressure is credited. Secondly, the calculated containment accident pressure for determining available NPSH is calculated in a way that underestimates this pressure. For example, the operation of the containment sprays is assumed even though they are not safety-related and therefore would not normally be credited in a safety analysis. Operation of the containment sprays significantly reduces the containment pressure which, in this case, is conservative. Credit is also taken for the transfer of heat from the containment atmosphere to various structures to further reduce the calculated containment pressure. Leakage from the containment at the technical specification limit, L_a, is also assumed. The NPSH calculations also overestimate the temperature of the suppression pool water, an important factor in NPSH calculations. The ultimate heat sink temperature is assumed to be at the maximum value allowed by technical specifications. This limits the amount of heat which can be transferred from the suppression pool and maximizes the suppression pool temperature.

The rationale for not crediting containment accident pressure, according to Regulatory Guide 1.1 is (1) the possibility of "impaired containment integrity"[6] or (2) excessive operation of the heat removal systems (sprays) resulting in a pressure less than that needed to maintain an adequate NPSH margin.

Containment integrity in a Mark I pressure suppression containment such as Vermont Yankee's is continuously monitored during normal operation since the containment is inerted, that is, air is removed and the containment is filled with nitrogen gas. Any significant increase in the amount of nitrogen that must be supplied to the containment might be a sign of degradation in containment integrity and would be observed by the reactor operators. The operators would then take the appropriate action in accordance with the Vermont Yankee technical specifications. Another sign of loss of integrity would be the presence of oxygen gas in containment. Monitors provide continuous assurance that no oxygen is present. Again, if oxygen were detected, the operators would take the appropriate action in accordance with the Vermont Yankee technical specifications.

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[6] Regulatory Guide 1.1 (Safety Guide 1) "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," USNRC November 2, 1970

BWR emergency operating procedures, including Vermont Yankee's, include a note to the reactor operators to terminate the containment sprays before the containment pressure drops below that required for adequate NPSH.

The present Vermont Yankee ECCS and containment licensing basis, as with all licensed nuclear power plants, is derived from design basis accident analyses. The determination of available NPSH is based on design basis analysis. Design basis accidents are accidents postulated to establish limits on operating conditions and safety-related equipment requirements given in the technical specifications. The assumptions used in design basis analyses are chosen to reasonably bound expected conditions. Thus, as explained above, flows, temperatures, pressures, power, etc. bound the expected conditions at which available NPSH is important to safety.

If realistic, rather than pessimistic and bounding assumptions were used in the design basis safety analyses, credit for containment accident pressure might not be necessary and, in any case, the containment accident pressure required for available NPSH would be much less than predicted under the conservative assumptions.

The NRC staff has also considered the impact of credit for containment accident pressure for so-called beyond-design basis accidents during which the ECCS or the containment heat removal system may be called upon to function. For station blackout, anticipated transients without scram (ATWS) and Appendix R postulated fires, the suppression pool conditions are less severe than those for the design basis LOCA and credit for containment accident pressure is not needed. Credit for containment accident pressure is also not needed for shutdown conditions. For a suppression pool bypass event, the suppression pool would not be heated and NPSH would not be an issue. In addition, for these postulated events, no debris would be generated and therefore the flow losses are considerably lower for these events than for the design basis LOCA and the available NPSH consequently greater.

For these reasons the NRC staff considers it acceptable to credit a limited amount of containment accident pressure in determining available NPSH.

> 2) *Does the agency believe that it is <u>necessary</u> to operate at extended uprated power level, thereby creating the necessity for allowing credit for containment accident pressure? If the answer is in the affirmative, please identify the reason the agency thinks operating at extended uprated power level is necessary?*

Per discussion with the Vermont Yankee project manager, DLPM will prepare the response to this question.

> 3) *What is the agency's policy regarding review to draft (rather than final) review guidance?*

As discussed above, the NRC is not using draft guidance. The guidance being referred to was developed and was used for the review of industry responses to GL 97-04. DG 1107 (now Regulatory Guide 1.82 Revision 3) was discussed with the NRC Committee to Review Generic Requirements (CRGR), ACRS and has received public comment. Since DG 1107 has now been issued by the NRC as Regulatory Guide 1.82 Revision 3, the guidance is no longer, in any case, draft guidance.

b. *Regulatory Position 2.1.1.2 of DG 1107 (at 16) states:*

For certain operating reactors for which the design cannot be practicably altered, compliance with Regulatory Position 2.1.1.1 [i.e., no credit for containment accident pressure] may not be possible.

Does the agency consider operation at OLTP to be a practicable alteration to allow compliance with Regulatory Position 2.1.1.1?

c. *At what uprated power level could Vermont Yankee operate and not claim credit for containment accident pressure in its NPSH calculations.*

The NRC staff has not performed calculations to determine the power at which credit for containment pressure is not required when using conservative assumptions.

As stated above, the NRC does not consider operation at OLTP to be a "practicable alteration" when evaluation and analysis have demonstrated that the credited pressure will be available for the design basis accidents, and when examined from a broader safety perspective than just the design basis accidents, an acceptable level of safety is maintained.

d. *Could you please identify for which licensees you have found it necessary to allow credit for containment accident pressure, and the reasons you found it necessary?*

The NRC does not maintain a list of plants for which credit for containment accident pressure has been approved, but the following list is believed to be reasonably complete. The reason in every case is that the as-installed pumps were found to need this credit when accident conditions following a LOCA are calculated conservatively. Increase in licensed power is not the context for crediting containment accident pressure in all cases.

Beaver Valley Unit 1 (PWR)

Browns Ferry Units 2 and 3 (BWR Mark I Containment)

Cooper (BWR Mark I Containment)

Dresden Units 2 and 3 (BWR Mark I Containment)

Duane Arnold (BWR Mark I Containment)

FitzPatrick (BWR Mark I Containment)

Fort Calhoun (PWR)

Hatch Units 1 and 2 (BWR Mark I Containment)

Monticello (BWR Mark I Containment)

North Anna Units 1 and 2 (PWRs)

Oconee Units 1, 2 and 3 (PWRs)

Oyster Creek (BWR Mark I Containment)

Peach Bottom Units 2 and 3 (BWR Mark I Containment)

Pilgrim (BWR Mark I Containment)

Quad Cities Units 1 and 2 (BWR Mark I Containment)

Surry Units 1 and 2 (PWRs)

e. *VY PUSAR Table 4-2 and Figure 4-6 identify that containment accident pressure credit is taken for a period of over two days after the accident. Since this constitutes the use of the reactor containment in a new manner, i.e., as an engineered safety feature to guarantee a minimum level of pressure over a 50 hour period, is additional containment pressure testing required to demonstrate pressure will be maintained for that period?*

The Vermont Yankee reactor containment already serves as an engineered safety feature. It serves as a pressure barrier to minimize leakage. Tests are done, as specified in the Vermont Yankee Technical Specifications, in compliance with 10 CFR Part 50 Appendix J, to ensure the pressure retaining capability of the containment. These tests verify compliance with a stringent leakage rate limit. In addition, as discussed above, the inerting of the containment provides several indications that containment integrity is continually maintained.

f. *What is the safety implication if credit for containment accident pressure is allowed? What is the agency's basis for allowing the regulatory requirement changed* [sic] *proposed by DG 1107?*

There is no safety implication. As explained under 2a1 above, credit for containment accident pressure is allowed under limited circumstances and only when the safety review determines adequate safety is maintained.